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Low recycling regime in ITER and the LiWall concept for its divertor [1]

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Abstract

The low recycling regime, although never considered as an option for ITER, may suggest a solution to its important issues, such as edge-localized modes, plasma and particle control, tritium inventory, damage of plasma facing components and dust accumulation, in a way consistent with both the ITER mission (including the ignition) and its baseline design and safety. Such a regime can be approached using liquid lithium surfaces efficiently pumping hydrogen isotopes. An active area of about 40 m^2, covered by $\simeq 0.1$ mm thick lithium, which is replenished with the rate of 10 kg/hour would be capable of absorption of plasma D and T particles and at the same time consistent with the ITER limitations regarding lithium. For low recycling conditions, a new consideration is outlined for the helium ash pumping problem.

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1 Introduction

The ITER device, targeting the fusion gain factor Q of 10 and the fusion power of 0.4-0.5 GW, has outstanding PSI issues. The power flux to the divertor plates is estimated to be 5-10 MW/m^2, while the peak flux due to the edge localized modes (ELMs) can significantly amplify this flux by a number, which cannot be determined reliably from the present day experiments. Not only the power, but also the energy of the plasma ions hitting the divertor plates is important because of sputtering of the plate material. ITER will address, for the first time in any machine, the problem of the helium ash exhaust and the tritium cycle issues.

The conventional approach to the plasma facing components (PFC) relies on cooling down the plasma, as it approaches the divertor plates, by enhancing its density. In addition to reduction of ion energy, this also enhances the radiation, which is beneficial for reduction of the power load to the PFC.

Although, seemingly self-consistent, this approach has several drawbacks in relying on complicated, difficult to scale phenomena. First, the low edge temperature plasma with substantial core heating exhibits ELMs, which deteriorate the performance and temporarily destroy the desirable state of the scrape off layer (SOL).

Secondly, cooling down the plasma edge automatically leads to the temperature gradient in the plasma core and to ion-temperature gradient turbulence. This instability is the major mechanism of loss of the thermal energy from the plasma and one of the reasons why reaching ignition is problematic even in large devices, like former ITER-EDA (Dimits at al , 2000).

Concerning PFC's, the insufficient confinement for ignition requires extra power from external sources, which not only reduces the Q-factor, but also leads to additional load (0.5 of the α-particle power in ITER) on the material surfaces.

Thirdly, the cooling down of the plasma edge by additional gas puff not only increases the load on the pumping system, but also pumps tritium, delivered to the plasma by the pellet injection. This makes the tritium fueling regime essentially "low" recycling.

This paper discusses an alternative approach, based on use of lithium (Li) plasma facing surfaces and a low recycling regime. Referred to here as the "LiWall" regime, it assumes low recycling for both ions and electrons (i.e., suppression of the secondary electron emission). While not being explored at the same degree as the conventional one, this regime would significantly simplify the physics of plasma confinement and stability, and make them predictable and scalable from the small to large experiments.

Only a conceptual level is addressed here. Sect.2 provides the basics of confinement and stability. Earlier (Krasheninnikov et al, 2003), it was shown that ITER would be ignited if the LiWall regime will be achieved. Sect.3 and Sect.4 discuss the potential options for a LiWall divertor and a helium pumping scheme. Suppression of the secondary electron emission is left out of the scope of the paper.

2 Confinement and stability of the LiWall regime

The experiments on TFTR, T-11, and CDX-U Majeski et al (2004) have shown the ability of lithium coated surfaces to pump hydrogen isotope ions from the plasma and provide low recycling boundary conditions and density control. The idea of the LiWall regime goes further than this in suggesting also the necessity to suppress the secondary electron emission. Both conditions can be expressed as

$$\Gamma^{micro,ions}_{edge\rightarrow wall} \simeq \Gamma^{ions}_{convective}, \quad \Gamma^{micro,electrons}_{edge\rightarrow wall} \simeq \Gamma^{electrons}_{convective}. \tag{1}$$

Here, $\Gamma^{micro,ions/electrons}_{edge\rightarrow wall}$ is the partial flux of plasma species to the wall, while $\Gamma_{convective}$ is the particle flux from the core of the plasma. With perfectly pumping walls, these two fluxes would be equal.

If combined with the third requirement of the LiWall regime, i.e., of the core fueling, the plasma edge temperature T_{edge} will be automatically high and comparable to the central temperature T_{core}

$$T_{edge} = \frac{1}{5\Gamma}\int P_{heat} dV \simeq T_{core}, \quad \Gamma \equiv \Gamma^{micro,ions}_{edge\rightarrow wall} + \Gamma^{micro,electrons}_{edge\rightarrow wall}. \tag{2}$$

The factor 5 in this equation is for Maxwellian distribution, which for good confinement would be close to reality. Deviations from a Maxwellian plasma will still leave the temperature at the edge comparable with its core value.

Fig.1 explains the two confinement regions in the plasma core, if the deposition of the fueling is localized at some distance from the plasma boundary. Deeper in the core, there is a conventional region, where the thermo-conduction determines the energy losses, while between the fuel deposition region and the plasma boundary there is a specific region, where energy is lost only together with the particles (rather than through thermo-conduction).

With the "perfect" pumping conditions at the boundary, the thermo-conduction losses are not significant, the turbulence is absent and confinement is expected to be much better than in the conventional plasma regimes. Simulations have

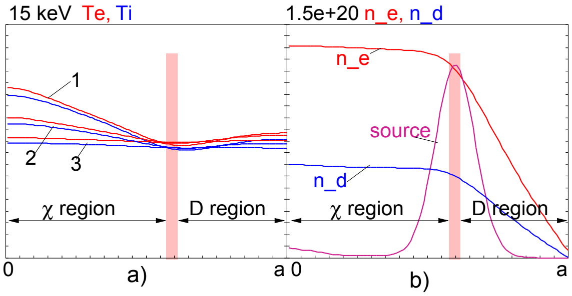


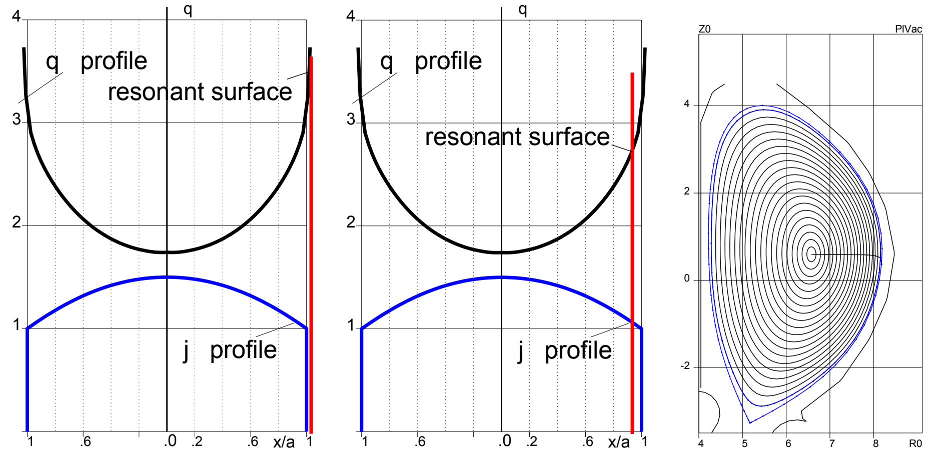
Fig. 1. χ- and D- confinement regions in the low recycling regime. (a) Electron and ion temperatures for three values of thermo-conduction coefficients. (b) Electron, ion density and the particle source. The curves 1,2,3 are calculated for one of standard models of thermo-conduction, and enhanced by a factor of 2 and 10, correspondingly.

shown that ITER would be ignited in the LiWall regime (Krasheninnikov et al, 2003)

LiWall regime suggests significant changes in the stability properties of the plasma. High edge plasma temperature would automatically lead to a finite current density at the last closed magnetic surface. At the same time, by controlling the low density at the plasma edge, good plasma pumping prevents the build up of the steep pressure gradient near the edge, despite its high temperature.

It is really remarkable that such a combination, contrary to the so-called "ballooning-peeling" concept, in fact, leads to a higher plasma stability and stabilization of ELMs.

In the case 1 in Fig.2, when a resonance surface with wave numbers m, n is possible just outside the plasma boundary, plasma is unstable (to the so-called peeling modes). In the case 2 in Fig.2, the modes, whose resonant surfaces are just inside the current density jump, are stabilized by the jump in j. The plasma, limited by the separatrix, always corresponds to the second case. *If the current density is finite at the last closed magnetic surface, peeling modes are not possible for this case.* Earlier (Medvedev et al, 2003), same conclusions



a) b) c)

Fig. 2. Current density j (blue curve) and the safety factor q profile (black curve) and position of the resonant surface (red line) relative to the j jump. a) Case 1: $mq_a < n$, ideally unstable, b) case 2: $mq_a > n$ tearing stable, c) LiWall + Separatrix: $q_a = \infty$ ideally & tearing stable

were made using the KINX code (for an ideal magneto-hydrodynamic model).

These two fundamental properties of the LiWall regime, i.e., enhanced confinement and enhanced stability, make it extremely attractive for the fusion oriented devices.

3 Conceptual Li-based options for the ITER divertor

While liquid lithium surface can not be damaged by the plasma particles, it looses its hydrogen pumping capabilities at the surface temperatures higher than $\simeq 400^0$ C. In high power devices, for the purposes of pumping of the plasma particles, the surfaces of the Li covered walls should be actively cooled. Also, it automatically means, that for relatively low thermo-conduction of Li (1 mm thick Li layer conducts 4.5 MW/m^2 of the heat flux at the temperature drop of 100^0 C across the layer) the Li layer should be very thin.

In fact, for ITER this requirement correlates with limitations on the overall inventory of Li inside the machine, while being consistent with the necessary pumping capacities. With $\simeq$ 10 atomic % of hydrogen consumption the 10 kg/hour rate of lithium replenishment would be sufficient for ITER.

In terms of Li inventory, e.g., the surface area of 40 m^2 (comparable with the surface of the ITER divertor plates) covered by a $h \simeq 0.1$ mm thick liquid lithium corresponds to 4 L (or 2 kg) of Li, certainly within the design limitations of $\simeq$ 30 kg. At the same time, even this layer would be excessive for the pumping purposes. Thus, the gravitational velocity V_g along the inclined surface (Li viscosity $\nu_{370^oC} \simeq 5 \cdot 10^{-4}$ and density $\rho \simeq 500$, $g = 9.8$) or Marangoni flow V_M (Guyon et al, 2001) due to the Li surface tension dependence on the temperature, $\frac{d\sigma(T)}{dT} = -1.62 \cdot 10^{-4}$, (all SI Units)

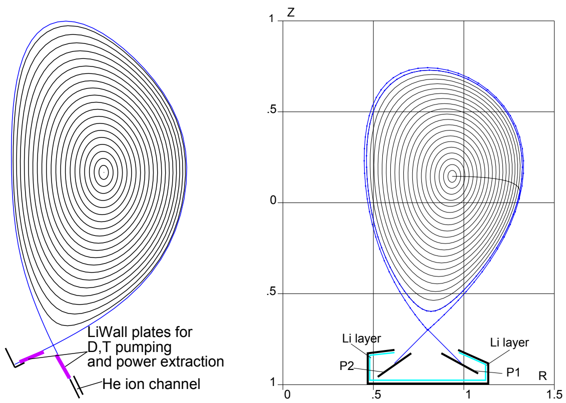
$$V_g = \frac{\rho g h^2}{2\nu} \sin\theta \simeq 0.048 \sin\theta, \quad V_M = \frac{d\sigma(T)}{dT} \frac{h \nabla T}{\nu} \simeq 0.0016 \tag{3}$$

(θ is the guide surface inclination angle) at $\Delta T \simeq 50^o$C will provide the higher than necessary replenishment rate (requiring $V < 1$ mm/sec). Al such low lithium speed, the intrinsic MHD effects due to poloidal or toroidal magnetic field are negligible.

Two options of using pumping Li wall surface are shown in Fig.3. One of them (Fig.3a) is similar to partial side walls and may be referred to as a Li "bleeding" bumper limiter. Its surface may be located at any place on the plasma surface (like a bumper limiter, divertor plates, etc). Due to the enhanced Larmor radius, the high edge plasma temperature, in fact, simplifies the alignment of the Li surface with the plasma, which is essential for this case.

The second idea (Fig.3b) contains lithium on the inner surface of a box around the SOL target plates. It may be referred as a "black slit" Li divertor. In this case, independent of the material of the plates the lithium surface will absorb the plasma particles, even though it is not directly exposed to the plasma.

The first option allows use of the divertor space for helium pumping (see the next section), but requires a good alignment. The second one is less sensitive to the alignment but occupies the divertor space.



a) b)

Fig. 3. Two concepts of the Li pumping of a separatrix limited plasma. a) Plasma cross-section with the partial Li side walls ("bleeding" bumper limiter); b) The "black slit" divertor with a lithium layer on the inner surface around the divertor target. P1 and P2 are two divertor plates.

4 Helium pumping at low recycling plasma edge conditions

The conventional approach for helium is a gas-dynamic scheme, when the high neutralized plasma pressure is created at the divertor plates. Then the mixture of D,T, and He flows through the pumping duct to the pumping volume.

This scheme is not applicable for the low recycling regime. Nevertheless, even with no ability to directly pump the helium, the LiWalls are, in fact, consistent with pumping helium as an ionized gas.

It is essential that in the LiWall regime all the power from the plasma is absorbed by the lithium, or by other surfaces. LiWalls separate the He ash from the plasma D,T particles. While D,T ions are absorbed by the lithium surface, the helium is released as a relatively cold neutral gas.

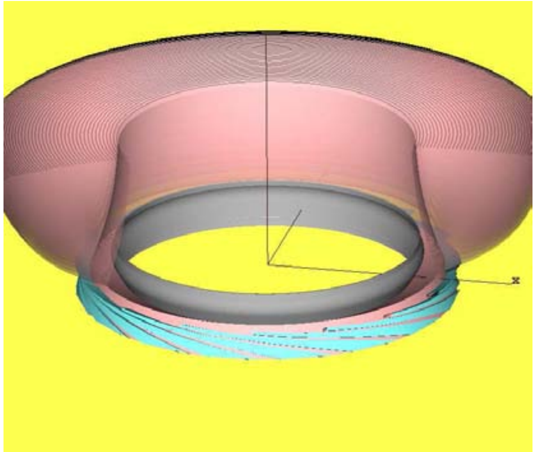
As a result, the collisionless conditions will be created for He, which, while migrating between material surfaces and the plasma boundary, will follow the field lines as soon as it is re-ionized near the SOL.

In the form of (low energy) ions, the He can be directed along the long "honeycomb" channels to a pumping volume. While the He ions do not interact with the channel walls (aligned with the magnetic field lines), the back flow of the neutral He from the pumping volume is suppressed by the friction on the walls.

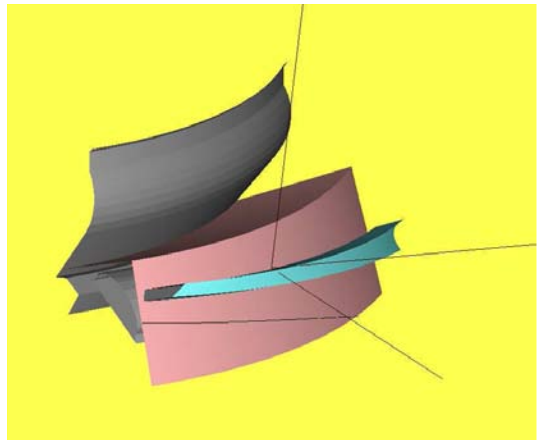
If d is the characteristic size of the channel cross-section and L is the length of the channel, even with the same temperature of He ions and He neutrals, the back flow is approximately d/L times smaller than the flow of ions.

Fig.4 shows the geometry of a conceptual He pumping duct with "honeycomb" channels situated at the outer leg of the ITER plasma separatrix. It is noticeable that for the typical tokamak condition, the poloidal magnetic field B_{pol} is much smaller than the toroidal B_{tor}. This allows for a short in the poloidal direction, but long in toroidal direction, honeycomb channels, highly resistant for the neutral gas flow.

For ITER the ratio of the poloidal extent of the duct to the length of the channels can be made as small as 1/20, thus leaving a substantial design space for the He pumping system. Good plasma control, possible in the LiWall regime, can contribute to the practicality of the described scheme.



a)



b)

Fig. 4. The "honeycomb" channel duct, calculated for ITER magnetic geometry. The size of channels is exaggerated. Also, B_{tor} is reduced by a factor of two. a) The toroidal He pumping duct. The number of channels was reduced for clarity; b) A separate channel, highly elongated along the field lines.

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